

NON-PUBLIC?: N
ACCESSION #: 9507110247
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Big Rock Point Plant PAGE: 1 OF 4

DOCKET NUMBER: 05000155

TITLE: MANUAL SCRAM DUE TO LOSS OF FEEDWATER
EVENT DATE: 11/29/94 LER #: 94-010-01 REPORT DATE: 07/03/95

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 054

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Michael D Bourassa, Licensing TELEPHONE: (616) 547-8244
Supervisor

COMPONENT FAILURE DESCRIPTION:
CAUSE: A SYSTEM: SD COMPONENT: MANUFACTURER:
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On November 29, 1994, reactor power escalation following a refueling outage was in progress. In addition, the second condensate pump was being returned to service following the replacement of a mechanical seal.

At 1530, the reactor was manually scrammed from about 54% power due to loss of suction pressure to the only operating feedwater pump. While filling and venting the condensate pump prior to operating, air was drawn through the common vent line to the operating condensate pump and discharged to the operating feedwater pump. All of the rods inserted and the plant functioned as designed. An Administrative cooldown limit for the Steam Drum was briefly exceeded, however an evaluation confirmed that engineering design limits were well within the specified parameters.

The loss of feedwater was caused by a procedure that had never been validated for the rare operating condition of returning a condensate pump

to service during power operation after it had been drained for maintenance. Corrective actions include procedure revision and validation for this operating condition.

END OF ABSTRACT

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IDENTIFICATION OF EVENT

Licensee shall report "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)..."

References

- a. 10 CFR 50.72(b)(2)(ii), and
- b. 10 CFR 50.73(a)(2)(iv).

CONDITIONS PRIOR TO THE EVENT

The reactor RCT! was operating at approximately 54% power. The control rods AA! were being withdrawn according to an approved sequence to facilitate power escalation. Both Recirculation Pumps AD;P!, the Number 1 Feedwater Pump SJ;P!, and the Number 1 Condensate Pump SD;P! were in operation. Feedwater control and Turbine Bypass Valve TRB;FCV! control were in automatic. System Operating Procedure (SOP) 15, Condensate System Hotwell to Reactor Feedwater Pumps, "Return the Condensate Pumps to Service" section, was being performed following mechanical seal SEAL! repairs to the Number 2 Condensate Pump.

DESCRIPTION OF THE EVENT

On November 29, 1994, Auxiliary Operators (AOs) were filling and venting the isolated (suction/discharge) Number 2 Condensate Pump prior to returning to service. (Draining, filling and venting is usually an outage activity, and rarely performed on line). When the equalizing vent VTV! to the isolated Number 2 Condensate Pump was cracked open, the air trapped in the pump was immediately drawn to the operating Number 1 Condensate Pump via the common vent line to the condenser COND! through the equalizing vent for the Number 1 Condensate Pump. The faltering Condensate Pump could not maintain the required suction pressure to the operating feedwater pump, causing it to automatically trip. A Control Room Operator (CO) recognized the condition and attempted to restart the

tripped feedwater pump. The first attempt failed. Two more attempts also failed. The low suction pressure condition could not be overcome due to the trapped air, the feedwater regulating valve SJ;FCV! automatically opening in response to low feedwater flow, and the opening of the condensate reject valve SD;LCV! on high hotwell level. At 1530, the reactor was manually scrammed due to the loss of suction pressure to the only operating feedwater pump. All of the rods inserted and the plant functioned as designed. Scram actions were completed by the operators. In addition, another attempt was made to restart the tripped feedwater pump. This time the pump remained running.

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At 2000, the Control Operators observed that a Final Hazards Summary Report cooldown limit of 100 degrees F per hour for the Steam Drum was briefly exceeded by an average of 24 degrees F per hour; however an evaluation confirmed that engineering design limits remained well within the specified parameters. (The steam drum will withstand a normal start-up and shutdown, 100 degrees F per hour from 100% power and 594 degrees F, approximately 2000 times; and approximately 100 emergency shutdowns from 100% power with a cooling rate of 384 degrees F per hour). Currently, the steam drum has accumulated 582 normal start-up and shutdown cycles, and zero emergency shutdown cycles.

ROOT CAUSE EVALUATION

The root cause of this event has been attributed to a less than adequate procedure. SOP 15, Condensate System Hotwell to Reactor Feedwater Pumps, Return the Condensate Pumps to Service section, has never been validated for the operating conditions experienced November 29, 1994. The loss of feedwater was caused by returning a condensate pump to service during power operation after it had been drained for maintenance and introducing air into the system. Draining, venting, and filling the condensate pump(s) to return to service is routinely performed during outages, not during normal power operation.

CORRECTIVE ACTION

Immediate

A daily order was written and posted on December 14, 1994, to prohibit the use of SOP 15, Condensate System Hotwell to Reactor Feedwater Pumps, Return the Condensate Pumps to Service section.

To Prevent Recurrence

1. SOP 15, Condensate System Hotwell to Reactor Feedwater Pumps, Return the Condensate Pumps to Service section, will be revised by January 15, 1996, to prohibit draining, venting and filling, of a condensate pump during normal power operation.

SAFETY SIGNIFICANCE

The consequences of this event were of minimal safety significance to the public and site employees because the reactor RCT! was placed in a safe shutdown condition. All systems required for safe shutdown were available and operated as designed. No Engineered Safeguards Features (ESFs) were actuated following the manual Reactor Protection System (RPS) actuation.

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SAFETY SIGNIFICANCE (Continued)

The Loss of Feedwater event is analyzed in the Updated Final Hazards Summary Report, therefore the plant was never in an unanalyzed condition. In the event of a total loss of feedwater, core cooling can be accomplished by the Control Rod Drive Cooling Water pumps AA,P! and the Main Condenser. One Control Rod Drive pump is sufficient for decay heat removal. If the Control Rod Drive pumps and or the Main Condenser are not available, core cooling can be accomplished by the Emergency Condenser BL!. Should all of the above components fail or not be available, the Reactor Depressurization System and the Core Spray System BM! may be used for cooling the reactor core.

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